Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

FEB 20 2003

Docket No. 50-423
B18833

RE: 10 CFR 50.73(a)(2)(iv)(A)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Power Station, Unit No. 3
Licensee Event Report 2003-001-00
Millstone Unit No. 3 Reactor Trip Caused by Generator Backup Protection Fault

This letter forwards Licensee Event Report (LER) 2003-001-00, documenting an event, which occurred at Millstone Power Station, Unit No. 3, on December 23, 2002. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A) as any event or condition that resulted in manual or automatic actuation of any of the systems listed in 10 CFR 50.73(a)(2)(iv)(B).

There are no regulatory commitments contained within this letter.

Should you have any questions regarding this submittal, please contact Mr. Paul Willoughby at (860) 447-1791, extension 3655.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

Stephen P. Sarver, Director
Nuclear Station Operations and Maintenance

Attachment (1): LER 2003-001-00

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Unit No. 3
Millstone Senior Resident Inspector

IE22

Docket No. 50-423
B18833

Attachment 1

Millstone Power Station, Unit No. 3

LER 2003-001-00

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

Estimated burden per response to comply with this mandatory information collection request 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Power Station - Unit No. 3	05000423	1 OF 3

TITLE (4)

Reactor Trip Caused by Generator Backup Protection Fault

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
12	23	2002	2003	– 001 –	00	02	20	2003	FACILITY NAME	DOCKET NUMBER 05000

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
OPERATING MODE (9)	1	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)	
POWER LEVEL (10)	100	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)	
		20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)	x	73.71(a)(4)	
		20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)	
		20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER	
		20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or	
		20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		In NRC Form 366A	
		20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)			
		20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)			
		20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)			

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Paul Willoughby, Supervisor, Licensing.	860-447-1791, Ext. 3655

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE).	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 23, 2002, at approximately 0200 with the plant at 100% power, a reactor trip occurred following a Generator fault. The appropriate alarms were received just prior to the trip. In addition, the trip signal opened a series of breakers and the Turbine and Reactor tripped sequentially. The Turbine Stop Valves closed within one second. The Steam Generator level shrinkage caused a low level in the Steam Generators starting the Auxiliary Feedwater System within 6 seconds, which was an expected occurrence. During and after the plant trip all main and auxiliary equipment functioned as designed and plant operators stabilized the plant in MODE 3.

The generator stator fault on phase A was found to be due to an arc to ground associated with the T-4 high voltage bushing. Specifically, stator cooling water was leaking through a pin-hole at the joint area (braze material) of the tubular lead conductor and the cooling water inlet fitting. The leaking cooling water traveled along the bushing body underneath the insulation package where the leaking coolant picked up contaminants (making it more conductive). The leaking contaminated water ultimately traversed the entire bushing to the point of contacting the housing enclosure (ground). Per vendor evaluation, the leak was due to a void/porosity issue with the braze material itself. This was most likely a fabrication deficiency that deteriorated over time.

The leak was repaired through re-brazing on the T-4 tubular lead connector and system integrity was confirmed through pressure testing. Other appropriate corrective actions are being addressed in accordance with the Millstone Corrective Action Program.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

1. Event Description

On December 23, 2002, at approximately 0200 with the plant at 100% power, a reactor trip occurred following a Generator [GEN] fault. The appropriate alarms were received just prior to the trip. In addition, the trip signal opened a series of breakers and the Turbine and Reactor tripped sequentially. The Turbine Stop Valves closed within one second. The Steam Generator [SG] level shrinkage caused a low level in the Steam Generators starting the Auxiliary Feedwater [BA] System (AFW) within 6 seconds, which was an expected occurrence. During and after the plant trip all main and auxiliary equipment functioned as designed and plant operators stabilized the plant in MODE 3.

The generator stator fault on phase A was found to be due to an arc to ground associated with the T-4 high voltage bushing (HVB). Specifically, stator cooling water was leaking through a pin-hole at the joint area (braze material) of the tubular lead conductor and the cooling water inlet fitting. The leaking cooling water traveled along the bushing body underneath the insulation package where the leaking coolant picked up contaminants (making it more conductive). The leaking contaminated water ultimately traversed the entire bushing to the point of contacting the housing enclosure (ground). Per vendor evaluation, the leak was due to a void/porosity issue with the braze material itself. This was most likely a fabrication deficiency that deteriorated over time.

Preventative maintenance testing, customarily performed during refueling outages, involves the completion of pressure and vacuum decay tests to monitor the generator bushing for leaks. The September 2002 outage (3RFO8) conducted both tests yet the T-4 leak was not detected. During the December 2002 forced outage a pressure test was completed to identify the location of the leak. It was not until the air pressure was increased to the test maximum (90 PSI, which is not a standard practice) that the T-4 leak was found.

The Millstone Unit No. 3 generator utilizes a Stator Liquid Cooling Monitoring System (SLMS) and a Liquid Level Detection System (LLDS) to monitor and detect water leaks during generator operations. SLMS works on Hydrogen intrusion and the bushing cooling water pressure is approximately 35 PSI higher than the hydrogen atmosphere of the enclosure. LLDS is used to detect gross water leaks that accumulate on the enclosure floor. Since the bushing cooling water pressure is maintained at about 35 PSI higher than the hydrogen atmosphere of the enclosure, SLMS could not sense such a leak. Also, because of the amount of bushing water contamination and the potential of the bushing, the arc was instantaneous such that LLDS would not have accumulated any liquid for potential leak detection prior to the trip.

The arc medium consisted of contaminated water (verdigris metallic oxide particles and oil), which is more conductive than pure stator cooling water. This contaminated water traversed along a high voltage bushing (13.8 KV) to ground resulting in an instantaneous flashover.

This event is being reported pursuant to 50.73(a)(2)(iv)(A) as any event or condition that resulted in manual or automatic actuation of any of the systems listed in 50.73(a)(2)(iv)(B). This includes Reactor Protection System actuation (RPS) and Auxiliary Feedwater System initiation.

2. Cause

The cause of this event (a generator stator fault on phase A) was determined be due to an arc to ground associated with the T-4 high voltage bushing. Specifically, stator cooling water was leaking through a pin-hole at the joint area (braze material) of the tubular lead conductor and the cooling water inlet fitting. The leaking cooling water traveled along the bushing body underneath the insulation package where the leaking coolant picked up contaminants (making it more conductive). The leaking contaminated water ultimately traversed the entire bushing to the point of contacting the housing enclosure (ground). Per vendor evaluation, the leak was due to a void/porosity issue with the braze material itself. This was most likely a fabrication deficiency that deteriorated over time.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

The Auxiliary Feedwater System initiation was caused by Steam Generator level shrinkage, which was an expected occurrence.

3. Assessment of Safety Consequences

The purpose of the Stator Water Cooling System is to remove heat in the main generator stator produced during the electrical generation process, and supply cooling-water to static rectifiers [RECT] used in the excitation system. It also functions to purify the cooling system, maintaining a very low level of electrical conductivity.

The porosity void of the conductor braze, which caused the leak, was probably an initial fabrication deficiency. In order to detect this situation prior to tripping the unit, extraordinary means as: (a) performing Helium Tests (the most sensitive leak test), (b) an air Pressure Test above standard test pressures on the bushing box side of the cooling system during each of the previous outages, or (c) stripping the extensive insulation package (significant task) for each bushing for visual inspection, would have been necessary. None of these three activities are standard industry practices. Review of industry operational experience showed that there has only been one other station trip associated with all other GE water cooled HVB type generators, nuclear and non-nuclear, over the last 10 years. Thus, an HVB leak is considered a low probability event.

A reactor trip occurred following the Generator fault. The appropriate alarms were received just prior to the trip. In addition, the trip signal opened a series of breakers and the Turbine and Reactor were tripped sequentially. The generator protection relay coordination performed as required for this fault condition. The Steam Generator level shrinkage caused a low level in the Steam Generators starting the Auxiliary Feedwater System within 6 seconds, which was an expected occurrence. During and after the plant trip all main and auxiliary equipment functioned as designed and plant operations stabilized the plant in MODE 3. Upon reactor trip, the required safety equipment functioned as expected and procedural compliance was maintained. Therefore, this event is considered to be of low safety significance.

4. Corrective Action

The leak was repaired through re-brazing on the T-4 tubular lead connector and system integrity was confirmed through pressure testing. Other appropriate corrective actions are being addressed in accordance with the Millstone Corrective Action Program.

5. Previous Occurrences

No previous similar events were identified.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].